UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 14, 2011

Mr. Mark J. Ajluni
Manager, Nuclear Licensing
Southern Nuclear Operating Company, Inc
40 Inverness Center Parkway
Birmingham, Alabama 35201

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT 1, SAFETY EVALUATION OF RELIEF REQUEST HNP-ISI-ALT-10, VERSION 1, FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL, TEMPORARY NON-CODE REPAIR OF SERVICE WATER PIPING (TAC NO. ME4253)

Dear Mr. Ajluni:

By letter to the U.S. Nuclear Regulatory Commission (NRC), dated July 16, 2010, as supplemented by letter dated July 20, 2010 (References 1 and 2, respectively), Southern Nuclear Operating Company, Inc. (the licensee) submitted Relief Request HNP-ISI-ALT-10 from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) at the Edwin I. Hatch Nuclear Plant, Unit 2. Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee proposed a temporary non-code repair to a leak occurring in the Hatch Nuclear Plant Unit 1 (Hatch-1) Service Water System.

Based on the review of the information the licensee provided, the NRC staff concludes that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety and the alternatives provide reasonable assurance of structural integrity. Therefore, the licensee's proposed temporary non-code repair is authorized in accordance with 10 CFR 10 50.55a(a)(3)(ii) until an ASME Section XI Code repair/replacement is performed during the Hatch-1 1R25 refueling outage or until the next cold shutdown of sufficient time to perform the repair/replacement, whichever comes first. The Hatch-1 1R25 refueling outage is currently scheduled for early spring 2012. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions concerning this matter, please contact Patrick Boyle at (301) 415-3936.

Sincerely,

Gloria Kulesa, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosure: Safety Evaluation

cc w/encls: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST HNP-ISI-ALT-10

REGARDING TEMPORARY NON-CODE REPAIR OF

PLANT SERVICE WATER PIPING

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NUMBERS 50-321

1.0 INTRODUCTION

By letter dated July 16, 2010 (Reference 1), as supplemented by letter dated July 20, 2010 (Reference 2), in response to the U.S. Nuclear Regulatory Commission's (NRC, the Commission) request for additional information (RAI) via phone conversation on July 19, 2010, Southern Nuclear Operating Company (SNC, the licensee), submitted Relief Request (RR) HNP-ISI-ALT-10 for Edwin I. Hatch, Unit 1 (Hatch-1). The licensee proposed to defer the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) repair of a degraded service water system piping at Hatch-1. The licensee requested relief because the proposed temporary non-code repair is not permitted on ASME Code Class 3 piping without prior relief from the NRC.

On July 7, 2010, a through-wall leak in the carbon steel elbow of the Hatch-1 plant service water (PSW) piping system was detected during shift walk-down. This elbow is located on the 30-inch dilution line upstream of the isolation valve and is physically located in the PSW valve pit in the river intake structure. The degraded line is moderate energy piping located in a section of Division II PSW that is normally stagnant and it provides dilution water when required. The PSW system piping supplies cooling water to both safety and nonsafety-related components.

The licensee requested the NRC authorization for the use of RR HNP-ISI-ALT-10 in order to support the temporary non-code repair of the through-wall pinhole leak at the elbow of the service water system piping at Hatch-1 without a plant shutdown. On July 21, 2010, the NRC staff granted verbal authorization for the use of RR HNP-ISI-ALT-10 at Hatch-1 for up to the end of refueling outage 1R25 which is scheduled to begin in early spring 2012, or up to the next cold shutdown of sufficient time to perform the ASME Code repair or replacement, whichever occurs first.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g) specifies that inservice inspection of nuclear power plant components shall be performed in accordance with the requirements of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). In 10 CFR 50.55a(g)(6)(i) it states that the Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest, given the consideration of the burden upon the licensee.

In 10 CFR 50.55a(a)(3) it states that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. In 10 CFR 50.55a(g)(5)(iii) it states that if the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in Section 50.4, information to support the determinations.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

ASME Code Class:	Class 3
Component:	30-inch nominal pipe size carbon steel 90 degree elbow
System:	Service Water System

The component for which relief is requested is listed below.

System / Component	**Nominal Wall Thickness Inch**	**Component Material**
Plant Service Water System / 30-inch nominal pipe size (NPS) carbon steel 90 degree elbow	0.375	Carbon Steel

3.2 Applicable Code Edition and Addenda

The code of record for the fourth 10-year inservice inspection (ISI) interval at Hatch-1 is the 2001 Edition through 2003 Addenda of the ASME Code, Section XI.

3.3 Applicable Code Requirement

The ASME Code Section XI, Article IWA-4000, "Repair or Replacement Activities," provides the requirements for performing repair or replacement activities on components and their supports. This is used whenever a flaw is discovered that does not meet the ASME Code, Section XI, requirements. To perform a repair or replacement activity, IWA-4412 of the 2001 Edition through 2003 Addenda of the ASME Code, Section XI, requires that defect removal shall be accomplished in accordance with the requirements of IWA-4420.

3.4 Proposed Alternative

In lieu of performing the ASME Code repair of through-wall leak in the carbon steel elbow of the Hatch-1 PSW piping, the licensee proposed to implement a temporary non-code repair in accordance with the guidance of NRC Generic Letter (GL) 90-05. This non-code repair is applicable until the next refueling outage or the next cold shutdown, whichever occurs first.

Hardship of Repair

The licensee stated that the PSW supplies cooling water to both nonsafety-related Balance of Plant (BOP) components and safety related components. The BOP components are isolated in case of an accident. The BOP heat load is much higher than that required of safety-related components. Performing an ASME Code repair of this location would require that Division II of PSW be shutdown. With a division of PSW out-of-service, the Hatch-1 Technical Specification (TS) 3.7.2, Condition E, requires that the PSW subsystem be restored to Operable status within 72 hours. However, considering the magnitude of the BOP heat load, it is predicted that removal of a division from cooling service for the time required to make the ASME Code repair would force a plant shutdown before the end of 72 hours.

Degradation Mechanism

The licensee stated that the cause of the leak is postulated to be localized pitting, possibly due to microbiologically influenced corrosion (MIC) attack. The MIC is the prime suspect due to the stagnant conditions in this section of piping.

Flaw Sizing

The licensee stated that diameter of the pinhole leak is approximately ¼-inch. The ultrasonic testing (UT) measurements of wall thickness were obtained around the area of the leak to better understand the scope of degradation. The UT thickness measurements were taken in approximately ¾-inch radial intervals from the center of the leak. The measurements showed that at ¾-inch away from the leak, the wall thickness was found to average 0.06-inch thick. At 1.5 inches from the center of the hole, the wall thickness was measured to average 0.26-inch thick which is above the allowable minimum wall thickness of 0.179-inch which was conservatively calculated based on the design pressure of 180 psig. The licensee stated that the maximum operating pressure is 140 psig.

The licensee stated that the wall thickness returns to a nominal wall thickness of 0.375-inch (or greater) at a radial distance of approximately 3 inches from the center of the leak. The wall thickness measurements showed that the degraded area appeared to extend uniformly in a radial pattern from the center of the leak, resulting in a cone shaped defect.

The licensee stated that additional UT measurements were taken on the elbow to determine if there were other areas of significant degradation. Sufficient scanning along the axis of the elbow and around the elbow was performed to provide reasonable assurance that there are not any additional significantly degraded areas in the elbow. One low spot near the elbow-to-pipe weld was measured as 0.177-inch thick, which is just below the allowable minimum wall thickness of 0.179-inch. Additional wall thickness readings were taken circumferentially around this low spot and all measurements returned to greater than 0.4-inch within 1-inch of the low spot. The licensee stated that this low spot is acceptable because the 0.179-inch allowable minimum wall thickness was conservatively calculated using the design pressure of 180 psig while the maximum operating pressure is 140 psig. Using the maximum operating pressure of 140 psig, the required minimum wall thickness was calculated as 0.14-inch thick. Therefore, the low spot is not required to have a specific flaw evaluation.

Proposed Temporary Non-Code Repair

For temporary non-code repair, the licensee stated that a 6-inch branch connection modification, using the design and examination requirements of ANSI 831.1, 2007 Edition, was made and welded over the leaking hole to encapsulate the leak. A 6-inch branch connection was selected because it would be welded over material with a remaining thickness well above the minimum wall thickness of 0.179-inch. The licensee stated that because the degraded material around the hole could not be removed with the system in operation, the defect was left in place.

The licensee stated that a full penetration weld was utilized for welding of the branch connection to the 30-inch pipe. Precaution was taken to keep the welding area dry from the water leaking from the pinhole. The 6-inch branch connection roughly centered over the leak while the leakage passed through the branch connection in a stream without wetting the weld area. Other compensatory actions (e.g., plugs, temporarily installed flow diverters) were also considered and used as necessary in case of water spray at the weld determined to be a problem.

The licensee stated that the welding was performed in accordance with Welding Procedure Specification S-1:1-0-1, Revision 1, which references Section IX of the ASME Code, 1983 Edition through 1985 Addenda, and 1989 Edition through 1999 Addenda. This specification requires a 60 °F minimum preheat which is currently met due to heating effect of the ambient water temperature in the elbow which is in excess of 85 °F. The procedure does not require a post-weld heat treatment because the nominal thickness of the materials is exempt from post-weld heat treatment in the construction code.

The licensee stated that welding was performed with water in the line and with the system pressurized to approximately 120 psig. This does not create any welding problems based on the following factors.

- Welding with water in a pipe is performed frequently in the industry and, as discussed above, the water temperature meets the 60 °F minimum preheat.

- Welding is performed on elbow base material that is approximately the nominal wall thickness or thicker. The measurements indicate that the welding is performed on thicknesses ranging from 0.371-inch to 0.459-inch.

- With the water in the system acting as a heat sink, the resulting heat affected zone of the elbow base material caused by the welding should be relatively shallow.

- Since only the inner 0.179-inch of the base material is required for pressure containment, welding on base material with thickness ranging from 0.371-inch to 0.459-inch would not be expected to encroach upon the ASME Code required minimum wall thickness and should have no impact on the load bearing capability of the elbow during welding process.

The licensee stated that the completed welds are visually examined in accordance with Hatch-1 Procedure 45QC-INS-004-0 to the acceptance criteria in B31.1 of the 2007 Edition of the ASME Code for Pressure Piping. Any unacceptable indications are repaired in accordance with the requirements in B31.1 of the 2007 Edition of the ASME Code for Pressure Piping. In addition, the licensee stated that the liquid penetrant examinations are performed in accordance with Hatch-1 Procedure no less than 48 hours after completion of the weld to ensure any delayed cracking would be detected. Furthermore, visual VT-2 examinations of system pressure test are performed per IWA-4540 requirements of the ASME Code, Section XI.

Flaw Evaluation

The licensee stated that because the PSW system is functioning in an operable but degraded condition, the following evaluations are addressed to ensure that the structural integrity of the subject degraded piping is being maintained and will continue to be maintained until the ASME Code repair is performed at the next refueling outage or the next cold shutdown, whichever occurs first.

The licensee stated that a flaw evaluation was conducted in accordance with Section 3.0 of ASME Code Case N-513-2 to evaluate the leak. The use of ASME Code Case N-513-2 is technically acceptable because the subject 30-inch elbow is considered as a bent pipe. The ASME Code Case N-513-2 flaw evaluation determined that the structural integrity is being maintained. The licensee stated that additionally, the requirements of NRC GL 90-05 using the through-wall flaw method for flaw evaluation were evaluated and determined to be acceptable.

The licensee stated that the added weight of the non-code repair by the branch connection modification was reviewed and determined that it did not impact the stress analysis calculations.

The licensee stated that the cause of the degradation is believed to be MIC-related, which is a slow process that does not produce rapid degradation of the piping. This leads to the assumption that if further degradation were to occur on this area of the 30-inch elbow, it would be minimal and gradual, and there would be sufficient time to perform the ASME Code repair or replacement until the next refueling outage or the next cold shutdown, whichever occurs first. This assumption is further justified by the fact that the piping with the degradation is original plant piping, and has been in service for approximately 35 years. This type of slow degradation provides reasonable assurance that the calculations and evaluations associated with the current degradation would remain valid until the ASME Code repair is performed at the next refueling outage or the next cold shutdown, whichever occurs first.

Augmented Examination

The licensee stated that the degraded piping is moderate energy piping and located in a section of Division II PSW that does not normally experience flow. Therefore, five other similar locations at Hatch-1 were chosen for the augmented volumetric examinations per NRC

GL 90-05 requirement. These five locations will be examined within 15 days of the discovery of the leak as specified by NRC GL 90-05. If any flaw is detected with a thickness less than the allowable minimum wall thickness, the scope expansion requirements of NRC GL 90-05 will be met.

Evaluation Approach

The licensee stated that the most conservative method for determining leakage is to assume a hole wherever the measurements were below the minimum wall thickness. This assumption resulted in an evaluation of a postulated 3-inch diameter hole in the elbow for the flow diversion assessment. A loss of inventory from a 3-inch diameter hole in Division II of the PSW system was evaluated against the design flows to safety-related components during loss of coolant accidents (LOCA). The results of this evaluation showed that with a 3-inch diameter hole in 30-inch line all the safety-related components would receive adequate PSW flow during LOCA. Therefore, with the worst-case leak due to loss of material from the existing location, the PSW system would still be capable of providing the required cooling to all relevant components.

The licensee stated that the location that the leak is spraying on was considered for impact on other components. The leak is currently spraying on a section of concrete wall with no other exposed components in the adjacent area. This shows that there would be no impact to other components due to a direct spray of water from the flaw. This information provides reasonable expectation that this condition would not affect ability of the PSW systems, or other components located in the valve pit, to perform as designed.

The licensee stated that with respect to the potential for flooding due to excessive leakage into the valve pit, the final safety analysis report (FSAR) section indicates that the valve pit is below the maximum theoretical flood level such that the automatic backwash feature of the PSW strainers may not be available in the event of a flood. However, the FSAR states, "The strainer is designed so that, even without backwashing and assuming a 90% [percent] clogging of the strainers, the strainer differential pressure is not less than 3 psig and the system flow is not retarded." This provides reasonable assurance that the components in the valve pit would be capable of performing the necessary design functions in the event of flooding. Therefore, the amount of leakage into the pit, and/or the ability of the nonsafety sump pumps to remove the water do not affect the operability determination of the PSW system.

The licensee stated that the following actions will be performed until the proposed temporary non-code repair is performed:

- Daily rounds will be performed to identify further degradation of the affected area as evidenced by a significant increase in the leakage rate. If a significant increase in leakage is detected, a UT examination will be performed to assure that the criteria used to evaluate the structural integrity remains valid.

- The area will be UT examined on a 30-day frequency to assure that unexpected degradation is not occurring and that the structural integrity of the elbow is being maintained.

The licensee stated that the following actions will be subsequently performed in the time period after the temporary non-code repair is made until the ASME Code, Section XI, repair or replacement is performed:

- Daily rounds will be performed to identify any signs that additional degradation is occurring.
- The area around the encapsulation will be ultrasonically examined on a 30-day frequency to assure that degradation outside of the encapsulated area is not occurring and that the structural integrity of the elbow is being maintained.

The licensee stated that based on the above actions the structural integrity of the PSW piping is being maintained and will continue to be maintained until the ASME Code repair or replacement is performed at the next refueling outage or the next cold shutdown, whichever occurs first.

3.5 Duration of Relief

RR HNP-ISI-ALT-10 is submitted for approval until an ASME Section XI Code repair/replacement is performed during the Hatch-1 1R25 refueling outage or until the next cold shutdown of sufficient time to perform the repair/replacement, whichever occurs first.

4.0 STAFF EVALUATION

The NRC staff notes that an ASME Code repair is required to address an identified through-wall flaw in an ASME Code Class 3 component to restore the structural integrity of the component independent of the operational mode of the facility. The ASME Code, Section XI, Article IWA-4000, provides the requirements for performing ASME Code repair or replacement on components and their supports. The NRC Inspection Manual, Part 9900: Technical Guidance, "Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety," provides guidance for determinations of operability and assessments of functionality for resolution of degraded or nonconforming conditions adverse to quality or safety. Section C.12 of NRC Inspection Manual, Part 9900, requires that if a leak is discovered in an ASME Code Class 1, 2, or 3 component in the conduct of an ISI, maintenance activity, or facility operation, corrective measures may require repair or replacement activities in accordance with Article IWA-4000 of Section XI of the ASME Code. NRC GL 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," provides a basis to allow temporary non-code repairs for ASME Code Class 1, 2, and 3 piping. The licensee proposed to follow the requirements in NRC GL 90-05.

On April 16, 2008, the NRC issued Revision 1 to the NRC Regulatory Issue Summary 2005-20, Revision to NRC Inspection Manual, Part 9900: Technical Guidance, "Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety." Section C.11 of the updated NRC Inspection Manual, Part 9900, states, in part, that the NRC staff accepts NRC GL 90-05 for conclusively establishing that a TS- required ASME Code Class 3 piping system that contains a flaw has adequate structural integrity. Section C.12 of the updated NRC Inspection Manual, Part 9900, states, in part, that the licensee may evaluate the structural integrity of ASME Code Class 3 piping by evaluating the flaw using the criteria of NRC GL 90-05, Enclosure 1, Section C.3.a.

Further, relief from the ASME Code requirements is needed even if the structural integrity is found acceptable when applying NRC GL 90-05.

NRC GL 90-05, Section C.1, states, in part, that the licensee can only use this provision if the flaw was detected during operation and an impracticality determination was made. For the purpose of this GL, impracticality is defined to exist if the flaw detected during plant operation is in a section of ASME Code Class 3 piping that cannot be isolated for completing a code repair within the time period permitted by the limiting condition for operation (LCO) of the affected system as specified in the plant TSs, and performance of code repair necessitates a plant shutdown. The licensee stated, during plant operation and shift walk-down, a pin-hole leak was discovered in the 30-inch diameter 90 degree elbow. The licensee determined that the leak was from a through-wall flaw located in a section of piping in the PSW system that cannot be isolated to complete ASME Code repair without a plant shutdown. Hatch-1 TSs provides only a limited 72-hour timeframe for this evolution. Therefore, the licensee concluded that it was impractical to perform Mid-Cycle repair. The NRC staff finds the licensee provided sufficient basis for impracticality of compliance. Therefore, the NRC staff finds the licensee met the impracticality requirement in accordance with NRC GL 90-05.

The NRC GL 90-05, Enclosure 1, Section C.2, states, in part, that a root cause determination and flaw characterization should be performed. The licensee determined that the root cause of the leak could be from localized pitting, possibly due to MIC attack. The licensee stated that MIC is the prime suspect due to the stagnant conditions in this section of piping. The licensee obtained UT measurements of wall thickness around the area of the leak to determine the scope of degradation. The licensee determined that wall thickness is at or above a nominal wall thickness at a radial distance of approximately 3 inches from the center of the pinhole leak. The licensee determined from the wall thickness measurements results that the degraded area appeared to extend uniformly in a radial pattern from the center of the leak, resulting in a cone shaped defect. The NRC staff finds the licensee root cause determination and flaw characterization are acceptable. Therefore, the NRC staff finds that the licensee met these evaluation requirements in accordance with NRC GL 90-05.

The NRC GL 90-05, Enclosure 1, Section C.3, requires the licensee to assess the structural integrity of component by performing a flaw evaluation. Section C12 of NRC Inspection Manual, Part 9900, states that GL 90-05, Enclosure 1, Section C.3, provides NRC acceptable flaw evaluation guideline for through-wall flaws. By letter dated July 20, 2010, the licensee in response to the NRC staff RAI submitted its flaw evaluation for the NRC review. The NRC staff reviewed the licensee's flaw evaluation against NRC GL 90-05 guidelines. The NRC staff finds the licensee's assumption (i.e. the pinhole leak had an opening diameter of 3 inches) was conservative. In addition, the NRC staff noted that the licensee satisfied the condition for acceptable flaw size specified in NRC GL 90-05, Enclosure 1, Section C.3, "through-wall flaw" approach, which specifies that the measured flaw length should not exceed either 3 inches or 15 percent of the length of the pipe circumference. Further, the staff agrees with the licensee's calculated stress intensity factor of approximately 32.216 ksi(in)$^{0.5}$ and finds it to be less than the critical stress intensity factor which represents the fracture toughness of the material. According to NRC GL 90-05, the critical stress intensity factor for ferritic steel is 35 ksi(in)$^{0.5}$. Therefore, the NRC staff finds that the licensee's flaw evaluation has satisfied the stress intensity factor requirements of NRC GL 90-05.

The NRC GL 90-05, Enclosure 1, Section C.4, requires the licensee to perform an augmented inspection once the flaw is detected and evaluated. The licensee stated that augmented inspection of five other similar locations at Hatch-1 will be performed. If any flaw is detected with a thickness less than the allowable minimum wall thickness, the scope expansion requirements of NRC GL 90-05 will be met. The NRC staff finds that the licensee has satisfied the scope expansion requirements of NRC GL 90-05, because it has examined five similar locations.

NRC GL 90-05, Enclosure 1, Section B3, requires the licensee to assess the integrity of the temporary non-code repair after the completion of the non-code repair by a suitable nondestructive examination method at least every 3 months until the ASME Code repair or replacement is performed. The licensee stated that daily rounds will be performed to identify any signs that additional degradation is occurring. In addition, the non-code repaired component will be UT examined every 30 days to ensure its structural integrity is being maintained.

The NRC staff has determined that the licensee has successfully met all requirements and taken all necessary actions in accordance with NRC GL 90-05 in support of deferring the ASME Code repair of a degraded service water system piping at its Hatch-1, until adequate time was available for the repair, but no later than the end of refueling outage 1R25, which is scheduled to begin in February 2012, or the next cold shutdown of sufficient time to perform the repair/replacement, whichever occurs first.

As discussed above, the licensee proposed the alternative on the basis of impracticality. The NRC staff notes that impracticality for the purpose of NRC GL 90-05, is defined to exists if the flaw detected during plant operation is in a section of ASME Code Class 3 piping that cannot be isolated for completing ASME Code repair within the time period permitted by the LCO of the affected system as specified in the plant TSs, and performance of ASME Code repair necessitates a plant shutdown. However, the NRC staff finds that shutting down the plant to perform Mid-Cycle repair would not be considered as impractical, but a hardship. The NRC staff would authorize the licensee's proposed alternatives under hardship or unusual difficulty without a compensating increase in the level of quality or safety pursuant to 10 CFR 50.55a(a)(3)(ii).

5.0 CONCLUSION

On the basis of its review, the NRC staff concludes that the proposed alternatives would provide reasonable assurance of operational readiness of the plant service water system and requiring an ASME Code repair immediately after determining the leak could have resulted in hardship or unusual difficulty without a compensating increase in the level of quality or safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the licensee's proposed alternatives until an ASME Section XI Code repair/replacement is performed during the Hatch-1 1R25 refueling outage or until the next cold shutdown of sufficient time to perform the repair/replacement, whichever occurs first. The Hatch-1 1R25 refueling outage is scheduled for February 2012.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

REFERENCES

1. Letter, M.J. Ajluni (SNC) To U.S. Nuclear Regulatory Commission containing ISI Program Alternative HNP-ISI-ALT-10 Version, Edwin I. Hatch Nuclear Plant, Unit 1, July 16, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102000178).

2. Letter, M.J. Ajluni (SNC) To U.S. Nuclear Regulatory Commission containing ISI Program Alternative HNP-ISI-ALT-10 Version 1, "Response to Request for Additional Information" Edwin I. Hatch Nuclear Plant, Unit 1, July 20, 2010 (ADAMS Accession No. ML102010574).

Principal Contributor: Ali Rezai, NRR/DCI

Date: January 14, 2011

If you have any questions concerning this matter, please contact Patrick Boyle at (301) 415-3936.

Sincerely,

/RA/

Gloria Kulesa, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosure: Safety Evaluation

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